SHIELDING DESIGN CONSIDERATIONS FOR A HIGH ENERGY NEUTRON SOURCE FACILITY

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SUMMARY

The design of a high energy neutron source facility to provide several radiation environments will require detailed calculations of the radiation shielding requirements for the protection of both personnel and equipment. In this work, the importance of the characterization of the radiation sources and of improved nuclear data are stressed. Also, the most relevant radiation shielding issues as well as guidelines for the various calculations that should be performed are discussed.

INTRODUCTION

During the FMIT[1,2] project several radiation shielding and activation calculations were performed. These calculations and related data are well documented and represent a substantial and valuable source of information.

From the standpoint of shielding analysis the availability of faster computers, enhanced computer programs, and larger nuclear data bases greatly facilitate today's calculations. New approaches in accelerator design which involve smaller deuteron losses and recent advances in engineering technology such as new materials, remote handling, and robotics will also simplify the shielding design. However, the calculational guidelines given in the FMIT literature are the bases upon which new calculations will be designed[3,4].

A facility based on the deuteron-lithium stripping reaction would provide high fluences of neutrons similar to that of a deuteron-tritium fusion environment for the testing of materials. The source intensity of approximately $4.00E + 16$ 14-Mev neutrons per second will provide fluxes on the order of $10^{15} n/cm^2 s^{-1}$ within several cubic centimeters of the test cell. However,the neutron source energy spectra has a tail that extends up to energies of 50 MeV and has a highly peaked forward directional distribution. Moreover, deuteron losses along the accelerator yield complex neutron and γ-ray radiation sources.

The objective of the high energy neutron source facility is to enhance the radiation damage to materials by exposing them to a much higher radiation fluence than that under which such materials would normally be exposed in a comercial fusion reactor. For this purpose, the FMIT and INGRID[5] designs required a deuteron beam of 100 – 150 mA with an energy of about 35 – 40 Mev to yield the necessary neutron source intensity.

Figure 1 shows a simplified scheme of a neutron source facility. In the injector room, deuterons from the ion source are accelerated and transported to the linac entrance. In the linac system radio frequency amplifiers (RF) are used to accelerate the deuteron beam to the desired energy level. After the last acceleration cavity in the linac system is the high-energy beam transport system (HEBT) wherein magnets are used to bend, direct, and focus the deuteron beam towards the lithium target in the test cell. The target consists of a continuous flow of liquid lithium which is approximately 2.5-cm thick and 10.0-cm wide. After irradiation in the test cell, the samples are tranported to the exam facility. Other areas of the plant include the control room, service and storage cells, offices, etc.

PRELIMINARY STUDIES

The first step in the realization of a radiation shielding assessment is the definition of the areas of concern — which in this case are the test cell, the linac system, the exam facility, the service, test support cells, etc. Levels of radiation allowed during operation and maintenance periods should be defined for each of these areas. It is also important to define radiation damage tolerances and temperature limits for equipment and other materials.

The utilization of advanced remote handling devices will certainly increase the overall plant availability and could also increase the radiation tolerance limits for the linac system, the test cell, and all areas that otherwise would be more heavily shielded or demand larger decay periods for hands-on access. Therefore, detailed studies of operation and maintenance procedures should be performed for each of the various areas of concern.

Two important and novel aspects of the facility relative to radiation shielding are the presence of neutrons with energies up to 50 MeV and activation due to $(d,\gamma X)$ reactions. It has been shown that deuteron activation can surpass secondary neutron activation in certain regions and for that reason measurements of deuteron activation were made for use in the empirical calculations for the FMIT facility[6]. Extensive studies[7,6] of the nuclear data required for neutron transport, activation, and heating calculations, were made for the FMIT and cross sections libraries for energies greater than 20 Mev were created. The approximations and models used for these cross sections should be checked to determine if further improvements are necessary. Lists of the most important radioactive nuclides and reactions that produced their parent nuclides can be found in the literature[7,5] and should be used as guidelines for similar work. This analysis will be very useful to select better materials according to their activation cross sections for the various radiation environments. The Radiation Shielding Information Center (RSIC) at ORNL can provide neutron cross section libraries for various materials for the energy range required.

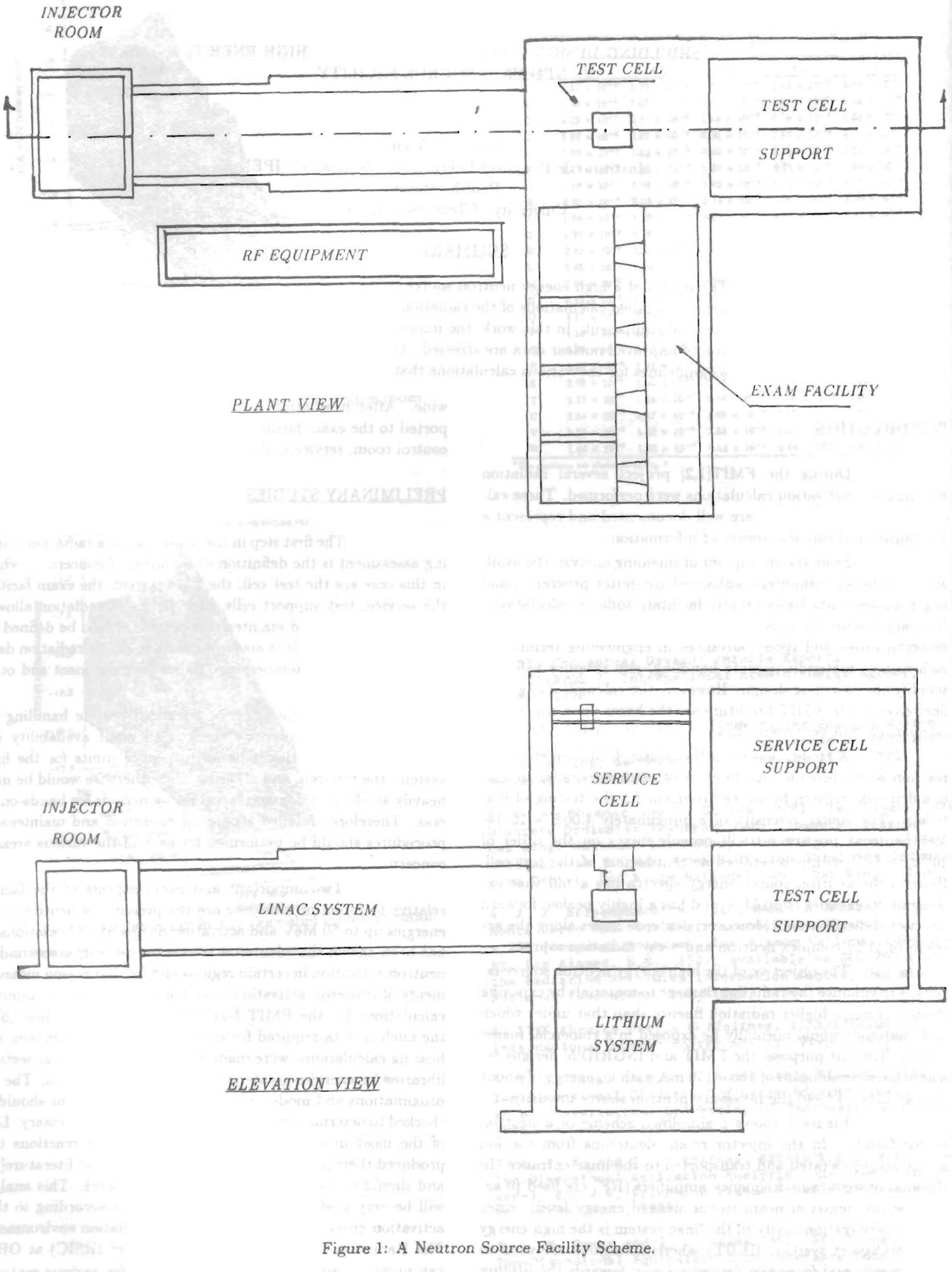

Figure 1: A Neutron Source Facility Scheme.

CALCULATIONAL GUIDELINES

Calculations[7,8] for the FMIT facility were performed almost exclusively with the MCNP[9] Monte Carlo code. However, some activation and heat deposition calculations benefit from the detailed descriptions provided by two-dimensional discrete ordinates codes. Also, the recent availability of three-dimensional discrete ordinates codes like TORT[10] may make the utilization of either method as only a matter of choice. Some of the more important calculations that should be performed are listed below.

1. Determination of shielding thickness and activation in the injector room.
   - It was assumed in the INGRID project that 150 mA of 350-keV deuterons would be lost during this stage. The deuteron concentration will increase rapidly within the stopping surfaces and the $d + d \longrightarrow {}^3He + n$ reaction will provide a substantial source of 2.45-MeV neutrons. Although a large number of deuterons are lost, the neutron yield for 350-keV deuterons is still very low compared with the yields at deuteron energies that occur in the final linac cavities. Therefore, the injector room will require much less effort in the radiation protection assessment. Monte Carlo and two-dimensional discrete ordinates transport codes can be used to determine environmental doses and an activation analysis will provide the decay times for safe acess by facility personnel.
2. Determination of the shielding thickness including deuteron and neutron activation for the linac vault.
   - Once deuterons achieve energies greater than the Coulomb barrier, which in the case of copper is 8.5 MeV, reactions of the type (d,nX) occur yielding high energy neutrons. High energy deuterons having large neutron yields with very high energies and a strong forward directional bias are produced. For the purpose of calculation, the last cavities best represent the radiation environment in the whole linac system. Because of the complexity of the source and the magnitudes of the dimensions involved, Monte Carlo codes such as MCNP[9] and MORSE[14] should be used to minimize thickness/cost of the concrete shielding. Points of concern are the neutron source description (position, energy, and direction distributions), the proper choice of materials that minimize activation (i.e. tantalum scrapers[5], boron loaded concrete[5], low manganese steel for the linac tank[15], the use of gold as plating material[15]), and the contribution of the ($d,\gamma X$) reactions.
3. Determination of shielding thickness including deuteron and neutron activation in the HEBT area.
   - Points of concern in this area are the eventual need for personnel access for manual adjustments of the focusing magnets, deuteron and neutron activation, and radiation streaming from the test cell.
4. Activation of air within the linac.
   - The most important reactions are reported by L. L. Carter et. al[7].
5. Determination of shielding thickness for the test cell walls.
   - Obviously the test cell has the most intense radiation environment in the facility. Neutrons produced by 40-MeV deuterons impinging on the lithium target have energies up to 50 MeV. One interesting feature of neutron spectrum is that it has a shoulder which begins at neutron energies of about 30 MeV. Studies in the literature[7] concluded that the influence of these neutrons in shielding calculations can not be ignored, in particular in the determination of the backwall thickness. Shielding calculations for the FMIT and INGRID test cells yielded concrete backwall thickness of several meters (6.0 and 4.4 meters respectively). The difficulties associated with such massive structures can be mitigated by underground construction which also will help to ease the radiation protection assessment.
6. Heat deposition within the test cell, bulk shield, and thermal shield (if used).
   - Because of the large amounts of energy deposition, there will be a need for cooling the concrete that faces the test cell as in the INGRID design or the utilization of a thermal shield as in the FMIT design. The beam energy which generates about 4 MW of power is almost entirely imparted to the lithium stream, which is cooled in the recirculating loop. The energy carried by the various radiations (neutrons and γ-rays) into the test cell will create heat sources in the surrounding material. The power associated with the flow of 4.00E+16 neutrons per second with an average energy of 14 MeV is 90 kW. Also, the heat generation in the first mesh along the beam axis in the concrete wall is 20 W/cm^3 — which was estimated by using the neutron and γ-ray fluxes obtained from a preliminary two-dimensional DORT[11] radiation transport calculation performed on a MFE Cray computer.
7. Neutron activation and radiation damage within the test cell.
   - Which includes test cell walls, test samples, and test cell equipment. Key issues here are the physical integrity of the walls and equipment for long-term operations, and the shielding requirements for test-sample transportation.
8. Radiation streaming through penetrations in the test cell.
   - Possible streaming paths towards the HEBT, the service and test support cells, and the exam facility should be investigated.
9. Neutron flux maps within the test volume.
   - One important set of data is the neutron generation cross sections[12] of the (d,Li) reaction. Detailed calculations of the neutron distribution in the test cell should be performed[13] using this set of data.
10. Test support, service, and service support cells.
    - Calculation of environmental doses. Points of concern are penetrations through the test cell and the activation of equipment. The utilization of remote handling for tasks in these areas is of major concern for the optimization of the plant operation.

11. Shielding and activation calculations for the lithium system.
    - The activity present in the lithium flow will require studies of the lay-out of equipment (pipes, pumps, etc) outside the bulk shielding of the test cell. Because of its own activity, proximity to the test cell, and difficulties in cleaning up residual activity from the pipes, the lithium circuit should be designed for long operational times.
12. RF and mechanical equipment areas adjacent to the linac vault.
    - Calculation of environmental doses.
13. The exam facility.
    - In the exam facility, the calculation of environmental dose is a function of the test-sample activation and radiation streaming through the test sample transport system. The ESNIT[3] proposal suggests a post-irradiation test facility and modular hot cells. These concepts can help maximize plant availability and influence other engineering design issues such as radioactive waste volumes.

CONCLUSIONS

It is necessary to achieve more information about the performance of materials under fusion-like environments so that comercial fusion energy can be available in the near future. Unlike other engineering issues, the shielding calculational experience accumulated so far by the nuclear fission industry can be readly applied to fusion related problems. The overall experience obtained in the design of accelerators during the past decade will make the design of a high energy neutron source possible without many of the difficulties experienced during the previous designs. The amount of data that was accumulated during the operation of the FMIT will greatly facilitate the design of a new plant regarding plant service lifetime, better defined flux characteristics, and particularly in the operation and maintenance of the plant — thereby enhancing plant availability and efficiency.

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REFERENCES

[1] E. W. Pottmeyer, Jr. The Fusion Materials Irradiation Test facility at Hanford. *Journal of Nuclear Materials*, 85 & 86:463–465, 1979.

[2] A. L. Trego, J. W. Hagan, E. K. Opperman, and R. J. Burke. Fusion Materials Irradiation Test Facility — a facility for fusion materials qualification. *Nuclear Technology/Fusion*, 4(2), 1983. Part 2.

[3] T. Kondo, H. Ohno, M. Mizumoto, and M. Odera. Selective energy neutron source based on the d-Li stripping reaction. *Journal of Fusion Energy*, 8(3/4):229–235, 1989.

[4] F. M. Mann. IFMIT an accelerator-based neutron source for fusion components irradiation materials testing capabilities. *Fusion Technology*, 15:295–297, 1989.

[5] M. J.Saltmarsh and R. E. Worsham. *INGRID - An Intense Neutron Generator for Radiation-Induced Damage Studies in the CTR Materials Program.* Technical Report ORNL-TM-5233, Oak Ridge National Laboratory, 1976.

[6] D. L. Johnson, F. M. Mann, and R. E. Sheuter. Measurements and evaluations of nuclear data to support early design needs of the FMIT facility. In *Symposium in Neutrons Cross Sections from 10 to 50 MeV*, page 495, 1980. DOE/NDC-21/L.

[7] L. L. Carter, R. J. Morford, and A. D. Wilcox. Nuclear data relevant to shield design of the FMIT facility. In *Symposium in Neutrons Cross Sections from 10 to 50 MeV*, page 431, 1980. DOE/NDC-21/L.

[8] L. L. Carter, R. J. Morford, A. D. Wilcox, and C. A. Rogers. Monte Carlo applications at Hanford Engineering Development Laboratory. In *A Review of the Theory and Applications of Monte Carlo Methods — Proceedings of a Seminar Workshop*, pages 7–30, Oak Ridge , Tennessee, APRIL 21-23 1980. ORNL/RSIC -44.

[9] Los Alamos Radiation Transport Group X–6. *MCNP - A Generalized Monte Carlo Code for Neutron and Photon Transport.* Los Alamos National Laboratory, September 1986.

[10] RSIC Computer Code Collection CCC–543. *TORT - Tri-Dimensional Discrete Ordinance Transport.* Oak Ridge National Laboratory, January 1990.

[11] RSIC Computer Code Collection CCC–484. *DORT - Two-Dimensional Discrete Ordinates Transport Code.* Oak Ridge National Laboratory, November 1988.

[12] D. L. Johnson and F. M. Mann. Measurements and calculations of neutron spectra from 35 Mev deuterons on thick lithium for the FMIT facility. *Journal of Nuclear Materials*, 85 & 86:467–471, 1979.

[13] W. P. Kelleher and G. L. Varsamis. Optimization of the testing volumes with respect to neutron flux levels in the two-target high flux d-Li neutron source for the International Fusion Materials Irradiation Facility. In *Proceedings of the IEEE Thirteenth Symposium on Fusion Engineering*, pages 674–678, October 2-6 1989. Knoxville, TN.

[14] RSIC Computer Code Collection CCC–474. *MORSE-CGA - A General Purpose Monte Carlo Multigroup Neutron and Gamma-Ray Transport Code System with Array Geometry Capability.* Oak Ridge National Laboratory, October 1987.

[15] R. J. Burke, A. A. Davis, and R. J. Morford. Radiation shielding issues on the FMIT. In *Ninth Symposium on Engineering Problems of Fusion Research*, pages 1916–1919, October 26,29 1981. Chigago, IL.